
Formerly Utilized MED/A Remedial Action

**Radiological Survey of the Building Site 421
Watertown Arsenal, V**

U.S. Dep
Assistant S
Division of Environm

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This is one of a series of reports resulting from a program initiated by the Atomic Energy Commission (AEC) for determination of the status of sites formerly utilized by the Manhattan Engineer District (MED) and AEC for work involving the handling of radioactive materials. Since the 1940's the control of over 100 sites that were no longer required for military programs has been returned to private industry or the public for unrestricted use. A search of MED and AEC records indicated that, for these sites, documentation was insufficient to determine whether or not decontamination work done at the time nuclear activities ceased is adequate under current guidelines.

This report contains the results of surveys of the current radioactivity at the Building Site 421, United States Arsenal Waterbury, Massachusetts. Findings of this survey indicate there are no significant areas involving an area of less than 6 000 cm² of identifiable low-level radioactivity on the concrete pad which is all that remains of Building 421.

The largest spot is approximately 5 000 cm². The other three spots are 1 000 cm² or less. The beta-gamma readings at these spots are 8.4 x 10⁴ dis/min-100 cm², 2.2 x 10⁵ dis/min-100 cm², 2.2 x 10⁵ dis/min-100 cm², and 1.1 x 10⁴ dis/min-100 cm². No alpha contamination was found at these locations. Gamma spectral analysis of a chip of contaminated concrete from one of the spots indicates that the contaminant is natural uranium. The contamination is "fixed" in the concrete and does not present an internal hazard.

er three spots indicated an elevated direct reading with the End
ector.

Radon daughter concentrations were determined at three locations
Building 421 pad. These were 0.00013 WL, 0.00011 WL and 0.00011 WL.
According to the Surgeon General's Guidelines found in 10 CFR 71.55, radon
daughter concentrations below 0.03 WL do not require remedial action in
structures other than private dwellings and schools.

Soil samples taken about the site indicate no elevated levels above
natural background levels in the soil. A gamma spectral analysis of a
sample obtained from the storm sewer line near the Building 421 pad
showed elevated radioactivity in the sample. It was therefore felt that
contamination is present in this sewer.

A radiological assessment was performed by the following Health
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Laboratory, Argonne, Illinois: R. A. Wynveen, W. H. Smith, C. E. Gray,
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BUILDING SITE 421
U. S. ARSENAL WATERTOWN,
WATERTOWN, MASSACHUSETTS

INTRODUCTION

During the Manhattan Engineer District/Atomic Energy Commission (M) era, an area known as Building 421 was occupied on the United States Arsenal Watertown Site. The site was used as a laboratory and office for the preparation and processing of uranium. Originally, laboratory work was conducted at Massachusetts Institute of Technology (MIT) and in 1945 the laboratory was moved to the Watertown Site. Since there are no records of a formal survey of the Building Site 421, it was decided to undertake a survey of the remaining floor pad of this building. All that is left of the Building 421 is a large concrete pad which had been the foundation of the building. The area of the pad was approximately 22 630 m². At the time of the survey, a portion of the pad was used as a storage area for concrete casings. The area covered by these vaults was approximately 5% of the total area available for survey. The arsenal property east of the Main Gate includes the Building 421 site, is at present owned by the Watertown Redevelopment Authority, Watertown, Massachusetts.

All accessible concrete pad areas were surveyed, with the exception of the area which was utilized for storage of the concrete vaults. In addition, the exterior south wall of Building 331, since it was adjacent to Building 330, was surveyed its entire length to the height of 2 meters. See Table 1A and 1B for locations of the areas surveyed. (All measurements were originally taken using the English system. The Standard International units are to the nearest approximation. For example, 2 inches are 5.08 cm.)

Instrumentation

Four types of survey instruments were used (Table 2). An Ekco Model FM-4G floor monitor, having a detection area of 325 cm², utilizing PAC-4G-3 electronics, was used to survey the pad. A PAC-4G-3 hand-held detector, 51 cm² sensitive area, was used to survey the vaults and other areas not accessible with the floor monitor. Double aluminum (0.85 mg/cm²) windows were used in both detectors. This allowed for low energy beta detection and increased instrument sensitivity. Both instruments were initially used in the beta mode. In the beta mode, the detector responds to alpha and beta particles and X and gamma rays. In areas where alpha activity was found which indicated a higher count rate than the instrument background, the beta reading was recorded and the instrument was then switched to the alpha mode and a reading of the alpha activity was recorded.

low Geiger-Mueller (GM) Detector, with an Eberline HP-190 probe was used to obtain a reading of the contaminated area at contact. This instrument was held one meter above the floor and was used to determine gross alpha and gamma background radiation levels throughout the surveyed area. In addition, a Health Physics Instrument Model HPI-1010, which is a low level integrating ion chamber, was used at selected locations about the perimeter to determine background readings. This instrument was used in the integrated mode in order to obtain an integrated reading from several locations. The HPI-1010 was calibrated using the gamma emissions from a Cobalt-60 (^{60}Co) calibration source. The End Window GM detector was calibrated using the gamma emissions from a Radium-226 (^{226}Ra) calibration source. The 4G-3 instruments were calibrated in the alpha mode using a flat disk source of infinitely thin Plutonium-239 (^{239}Pu) standard and in the beta mode using a flat plate, infinitely thin Strontium-90-Yttrium-90 (^{90}Sr - ^{90}Y) standard. The 4G-3 instruments were calibrated so as to provide an apparent activity in Becquerels (Bq).

It should be realized that the numerous isotopes which could be encountered will exhibit emission energies differing from that of ^{239}Pu and ^{90}Y utilized in the calibration. When detecting known isotopes that have alpha and beta energies differing from that of the standards, a conversion factor for the particular isotope is developed to determine the appropriate activity.

devices, was housed in the mobile laboratory, a converted Winnebago Home.

Smear Surveys

Dry smears were taken throughout the floor pad area of Building the outside wall of the adjacent Building 331. A standard smear is performed by applying moderate pressure by the tips of the first two fingers back of the filter paper. All smears were taken with No. 1 Whatman paper, 4.25 cm in diameter. Smears of 930 cm² were normally taken if an area or object was found which had a higher than normal ins background, a smear of 100 cm² was taken. A smear of 100 cm² was taken if an area indicated excessive dirt loading. The smears were taken in groups of ten using a 10-Wire Flat Plate Gas Proportional Detector developed at ANL, utilizing an Eberline Mini Scaler Model MS-2. At least one smear of each group was removed and counted in the more sensitive Measurement Corporation 2 π Internal Gas Flow Proportional Counter (PC-5) using a mylar spun top. (The mylar spun top is placed over the detector to negate the dielectric effect when counting samples on nonconducting materials such as paper.) This procedure was used as a more sensitive method for counting a selection of the smear samples. All smears of areas or objects with elevated direct readings, were counted in the PC Counter. In addition, any smears indicating readings above the instrument background level in the 10-Wire Assembly, were also counted in the PC-5 Counter. Smear

the instrumentation and smear count conversion factors used.

Table 1 includes the survey results, while the map in Figure 1A indicates the location of the smears. A number, n , indicates the location of the n th smear. Figure 1B indicates the smear locations of the areas of interest; R-1 through R-4 indicate the areas with an elevated concentration of radionuclides.

Samples

Air samples were collected using a particulate air sampler (PR-100) developed at ANL. (See Figure 1C for locations of air samples.) The collection medium consisted of a 200 cm² sheet of Hollingsworth & Lowry (70-0.23 mm) filter paper which collects the particulates present in the air. Flow rates of 15 or 40 cubic meters per hour (m³/h) were used. After collection, 5 cm in diameter, was removed from the filter media after collection and counted for both alpha and beta activity in the PC-5, utilizing a 10 cm² detector. The collection efficiency at these flow rates for 0.3 micron particles is approximately 99.9%. Sampling results were used to determine radionuclide concentrations and the presence of any long-lived activity. The calculations are presented in Appendix 2.

Samples

Soil corings were taken at selected locations in undisturbed areas in the immediate periphery of Building 421 to determine the deposition, if any, of radionuclides that could have been spilled or released from the site. The locations of the soil corings are indicated in Figure 1D.

corings were taken at each site to determine background levels

This was also done to identify any spurious anomalies that can be caused by surface contamination. Chemical (fluorometric) and gamma-ray spectral analyses were conducted on the soil samples.

The corings were effected using a 10.1 cm in diameter by 15.2 cm long right circular cylinder cutting tool, used as a golf-green hole cutter. The core was 30 cm in length and divided into four segments. Starting from the surface, three separate 5 cm segments are cut, bagged, and marked A, B, and C, respectively; the final segment of 15 cm was marked D. The purpose of the segmented coring is to determine what, if any, contamination might have occurred, to reduce the dilution of lower level soil with the upper level soil, and to reveal any overburden or back fill that may have occurred in the past years.

Eight soil corings were effected from the grounds adjacent to the 421 pad area. In addition, certain of these samples were chosen to be in the immediate vicinity of Building 34 and Building 41. Specifically, samples S-4 and S-5 were taken just south of the old Building 41 site and sample S-7 was taken just east of the old Building 41 site. The sampling area encompassed both the 421 and 41 sites since it was located between the two areas. Figure 1D depicts the soil sample locations.

All soil samples were processed at ANL (Figure 3) and shipped to a commercial laboratory (LEE Environmental Analysis Laboratory).

then drying for approximately 24 hours at 80° Centigrade. All samples were then reweighed, placed into mill jars (8.7ℓ), and milled until a sufficient amount of the soil sample could pass a No. 30 standard sieve. At no time were the rocks and heavy material ground or pulverized since this material would act as a diluent and, hence, lower the concentration of deposited material. After sufficient milling, the material was sieved using a No. 30 (No. 60) standard stainless sieve. The rocks and dross vs. sieved material were segregated, bagged, and weighed separately. Soil sample weights are given in Table 6. Aliquots of the sieved material were then loaded into polypropylene plastic containers. The amount varied according to the type of analysis to be performed, 100 grams for gamma spectral and radiochemical (fluorometric) analysis and 10 grams for radiochemical (fluorometric) analysis. Every effort was made throughout the sample preparation operation to avoid any possible cross contamination. Soil samples which were suspected of containing elevated amounts of radioactivity were processed in equipment separate from the soil samples considered to contain background levels. All equipment was scrubbed and air dried prior to the introduction of the next sample.

Water Sample

A water sample was obtained from the storm sewer which ran along the north side of the Building 421 concrete pad. Since it was a single pipe, only one sample was obtained. A 118 ml sample was taken

This section discusses the results of the survey and the findings. G-3 instrument readings and smear results were normalized to disintegrations per minute per one hundred square centimeters (dis/min-100 cm²) using the factors in Appendix 1 and are equated to natural uranium. Data is reported in net counts, i.e., the background count rates were subtracted from the gross count rates prior to converting from dis/min-100 cm² to c/min-100 cm². The beta mode readings are corrected to remove any contribution. The μ R background readings which were taken for various periods of time have been converted to μ R per hour. Table 3 summarizes the background reading for all modes of operation of the detectors and instruments used. The μ R readings taken about the pad are presented in Table 4 and their locations are depicted in Figure 1C.

The area accessible for survey on the floor pad was approximately 100 m² of the outside wall area of Building 331 up to 2 meters above the floor level.

Instrument Surveys

There were four areas on the concrete floor pad that indicated elevated background readings, R-1, R-2, R-3 and R-4. The locations of these areas are shown in Figure 1B. The beta-gamma readings at these four locations were 8.4×10^2 dis/min-100 cm², 2.2×10^5 dis/min-100 cm², 2.2×10^5 dis/min-100 cm², and 2.2×10^5 dis/min-100 cm².

Spots R-1, R-2 and R-3 were 100 cm² or less. Spot R-4 was found on the surface of the concrete and was approximately 5 000 cm².

The highest End Window reading was indicated at Spot R-2. The End Window reading was 0.09 mR/h with the instrument in contact with the concrete. Two of the other three spots indicated an elevated direct reading with the End Window Detector.

Eight background measurements using a Micro R meter were taken on the concrete floor pad. These μ R readings ranged from 4.9 μ R/h to 11.5 μ R/h. The range of μ R background readings in an uncontaminated area is 7-12 μ R/h. Two of the μ R readings taken on the concrete pad were found to be elevated.

A sample of spot R-2 was extracted and submitted for gamma spectroscopy analysis. Figure 2 shows the gamma spectrum that was observed using a Multi-channel Analyzer. Results of this analysis indicate the contaminant is natural uranium; therefore, the PAC-4G-3 instrument readings are equal to natural uranium.

No elevated readings were found on the outside wall of Building 421.
Area Surveys

Smears were randomly taken on the Building 421 floor pad and the Building 331 wall which is next to the Building 421 floor pad. Smears were also taken on the four spots of contamination found on the floor pad. No significant contamination was found.

in Appendix 2, the Radon-222 Working Level (WL) concentrations were determined at these locations. The results were 0.00013 WL, 0.00011 WL, and 0.00009 WL, respectively.

According to the Surgeon General's Guidelines found in 10 CFR 191.103, concentrations of radon daughters below 0.03 WL do not require remedial action in structures other than private dwellings or schools. (A copy of the Surgeon General's Guidelines is found in Appendix 6.)

Samples

Results submitted by LFE Environmental Analysis Laboratories, as listed in Table 8, are reported in picocuries per gram (pCi/g) for the Germanium (Ge) [Ge(Li)] spectral analysis and in micrograms per gram ($\mu\text{g/g}$) for the uranium fluorometric analysis. The latter concentrations were converted to pCi/g by means of the example calculation shown in Appendix 4.

The soil sample data along with background soil sample data is presented in Table 7. The background soil samples indicate natural uranium concentrations ranging from 0.9 to 4.8 pCi/g with one exception. The sample SM-2B which was 12.2 pCi/g. Since samples SM-1 and SM-2 were taken in close proximity, it is expected that their results should be similar. The elevated concentration of SM-2B may be a result of soil fertilization. Since elevated concentrations of uranium are present where phosphate-containing fertilizers are used, any fertilizing of the soil can result in an increased uranium concentration. Results of soil samples taken at the Watertown Arsenal are in the range of 0.9 to 4.8 pCi/g of uranium.

The water sample obtained in the storm sewer was counted on a Channel Analyzer. No radioactivity above background was detected in the sample.

Decontamination Guidelines

Both the draft ANSI Standard N13.12 for the "Control of Radioactive Contamination on Materials, Equipment, and Facilities to Be Released for Uncontrolled Use" and the NRC Guideline for "Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for By-Product, Source, or Special Nuclear Material", are considered as guidelines (see Appendix 5).

Since natural uranium was identified, the acceptable decontamination levels for natural uranium will be used. The NRC Guideline for natural uranium is as follows: the average is 5 000 dis/min-100 cm² of removable uranium is 15 000 dis/min-100 cm² α , and the removable is 1 000 dis/min-100 cm². The measurements for the average may not be averaged over less than one square meter and the maximum level applies to an area of not less than 100 cm². Also, the average and maximum radiation levels associated with the contamination resulting from beta-gamma emitters should not exceed 0.5 mrad/h at 1 cm and 1.0 mrad/h at 1 cm, respectively, measured through 7 mg/cm² of total absorber.

The ANSI Standard N13.12 is more limiting than the NRC Guideline for natural uranium. The allowable natural uranium activity is 5 000 dis/min-100 cm² of which only 1 000 dis/min-100 cm² can be removable. The

Standard is more restrictive in that its limits include both alpha activity and the NRC Guide includes only alpha activity. In the contamination found at Watertown Arsenal, beta-gamma activity is present but no alpha activity was detectable. It is important to note that a thin layer of dirt on the concrete pad could prevent the detection of alpha activity. If this occurred, an area of contamination would meet the NRC Guide but would exceed the ANSI Standard levels.

The areas of fixed contamination which were found to be greater than the 5 000 dis/min-100 cm² total for natural uranium as given in the ANSI Standard are depicted in Figure 1B and are as follows:

<u>Location</u>	<u>Counting Results</u>
R-2	2.2×10^5 dis/min-100 cm ²
R-3	2.2×10^5 dis/min-100 cm ²
R-4	8.5×10^4 dis/min-100 cm ² .

In an effort to evaluate the radiological health hazard associated with contaminated spots on the pad at the Watertown Arsenal, a credible hypothetical situation will be determined. One such situation which, of course, would involve the disturbance of contaminated concrete. A pneumatic hammering of the contaminated spot could result in a radioactive aerosol. This would appear to be a satisfactory form to consider for purposes of assessing a potential radiological hazard.

Even though the contamination found at R-4 is not the highest point on the area, this spot will be used to assess the radiological hazard since it contains the greatest level of total activity. The level of activity found at R-4 is 8.5×10^4 dis/min-100 cm² equated to natural uranium⁽¹⁾ (dis/min-100 cm²).⁽²⁾

When the sample from R-2 was taken by scraping the concrete with a trowel, the level of activity was decreased by approximately one half. It would appear that most of the activity at R-2 is confined to the top 0.6 cm of the concrete. But since the contamination at R-4 is not found in cracks or in the cement, it will be assumed that this level of 8.5×10^4 dis/min-100 cm² is constant through a depth of 10 cm and that this level of 8.5×10^4 dis/min-100 cm² is present in every 0.6 cm depth. This assumption will result in a conservative estimate of the internal exposure hazard.

cm² and the the depth of the jackhammering would be 10 cm, the tot

(A) involved would be as follows:

$$A = \frac{8.5 \times 10^4 \text{ dis/min}_{(nu)} \cdot 100 \text{ cm}^2}{0.6 \text{ cm}} \cdot 10 \text{ cm} \cdot 5\,000 \text{ cm}^2$$

$$= 7.1 \times 10^7 \text{ dis/min}_{(nu)}.$$

This is also equal to

$$A = 7.1 \times 10^7 \text{ dis/min}_{(nu)} \cdot \frac{1 \mu\text{Ci}_{(nu)}}{2.22 \times 10^6 \text{ dis/min}_{(nu)}}$$

$$= 3.2 \times 10^1 \mu\text{Ci}_{(nu)}.$$

Assuming the depth of the concrete cut from the jackhammer is 10 cm, the volume of concrete displaced (B) would be as follows:

$$B = 10 \text{ cm} \cdot 5\,000 \text{ cm}^2 = 5 \times 10^4 \text{ cm}^3.$$

The activity per unit volume of concrete (C) would be as follows:

$$C = \frac{3.2 \times 10^1 \mu\text{Ci}_{(nu)}}{5 \times 10^4 \text{ cm}^3} = \frac{6.4 \times 10^{-4} \mu\text{Ci}_{(nu)}}{\text{cm}^3}.$$

Since jackhammering results mainly in large pieces of concrete rather than small particles, the assumption will be made that only 10% of the

tivity per gram of total concrete suspended in the air (D) would

$$\frac{2.1 \times 10^{-4} \mu\text{Ci}_{(\text{nu})}}{\text{cm}^3} \cdot \frac{\text{cm}^3}{3\text{g}} = \frac{2.1 \times 10^{-4} \mu\text{Ci}_{(\text{nu})}}{\text{g}}$$

one can assume that the concrete is dry and that a maximum dust burden would be created from the jackhammer disturbance.

This maximum dust burden would, however, most probably be no greater than the Threshold Limit Value (TLV) for Nuisance Particulates. The TLV is $1.5 \times 10^{-2} \text{ g/m}^3$ of air or $1.5 \times 10^{-2} \text{ g/m}^3$ (3).

If an aerosol was generated which contained this TLV, the concentration of radionuclide (E) in the air would be:

$$\frac{2.1 \times 10^{-4} \mu\text{Ci}_{(\text{nu})}}{\text{g}} \cdot \frac{1.5 \times 10^{-2} \text{ g}}{\text{m}^3} = 3.2 \times 10^{-6} \mu\text{Ci}_{(\text{nu})}/\text{m}^3$$

$$3.2 \times 10^{-12} \mu\text{Ci}_{(\text{nu})}/\text{cm}^3.$$

It should be realized that only a limited number of people could be involved in such a postulated environment for any length of time.

Under these conditions, the level of concrete dust described herein would probably

be within the limits of a person's breathing zone due to the high density

In the preceding situation we have generated an aerosol 1.6

Since the jackhammering of the small area would not take the particles would soon fall out of suspension, the aerosol could last only 10 minutes. A person involved in this operation of time would inhale the following level of activity (G):

$$G = \frac{1.2 \text{ m}^3 \text{ of air}}{60 \text{ min}} \cdot 10 \text{ min} \cdot \frac{3.2 \times 10^{-6} \mu\text{Ci}_{(\text{nu})}}{\text{m}^3} = 6.4 \times 10^{-7} \mu\text{Ci}_{(\text{nu})}$$

The fraction reaching the organ of reference, which in this case are the kidneys, would be $0.028^{(6)}$. Therefore, the total reaching the kidneys would be:

$$H = 6.4 \times 10^{-7} \mu\text{Ci}_{(\text{nu})} \cdot 0.028 = 1.8 \times 10^{-8} \mu\text{Ci}_{(\text{nu})}$$

The maximum permissible burden of uranium for the kidneys of a human body, $[q(k)]$, is $5 \times 10^{-3} \mu\text{Ci}_{(\text{nu})}^{(7)}$. Comparing the total activity in the kidneys to the maximum permissible kidney burden the following ratio is obtained:

$$I = \frac{1.8 \times 10^{-8} \mu\text{Ci}_{(\text{nu})}}{5 \times 10^{-3}} = 3.6 \times 10^{-6}$$

...s, the person would receive 3.6×10^{-4} of a kidney burden
ration.

Even though these calculations are based on reasonable hypotheses, it must be realized that the actual activity could be greater than the calculated value. Simultaneous use of the jackhammer at the other contaminated areas would not significantly increase the hazard.

Only the GM End Window reading taken in contact with the contaminated surface was found to be greater than the instrument background. The reading was 0.09 mR/h. None of the GM End Window readings taken from the contaminated spots were distinguishable from the normal instrument background range of 0.03 to 0.05 mR/h. Therefore, no external hazard is envisioned.

from ^{235}U . This equals 4.5×10^{12} dis/min per Curie of natural uranium.
A standard Curie is 2.22×10^{12} dis/min.

Density of Elements and Common Material, as given in the Radiological Health Handbook, January, 1970.

The Threshold Limit Value is 15 mg/m^3 for Nuisance Particulates from the Occupational Safety and Health Standards subpart Z - Toxic and Hazardous Substances, May 28, 1975.

The Maximum Permissible Concentration values are given in "Standards for Protection Against Radiation," Code of Federal Regulations, Title 10, Part 20, Appendix B (April 30, 1975) (10 CFR 20), for an insoluble radionuclide in air in an uncontrolled area.

Specifications for Standard Man as given in the Radiological Health Handbook, January, 1970.

Report of Committee II of Permissible Dose from Internal Radiation (1959) International Committee on Radiological Protection (ICRP).

This is the maximum permissible kidney burden for ^{238}U as found in the Report of Committee II of Permissible Dose from Internal Radiation (1959) International Committee on Radiological Protection (ICRP).

A comprehensive radiological survey was completed at the Watertown, Watertown, Massachusetts. This site was used as and office areas for the preparation and processing of uranium.

Direct instrument survey of the pad of Building 421 and exterior wall of Building 331 which is adjacent to the pad, indicated areas of contamination on the Building 421 pad. These four spots, R-1, R-2, R-3 and R-4 on the maps, consist of the following contamination: 8.4×10^2 dis/min-100 cm² beta-gamma, 2.2×10^5 dis/min-100 cm² beta-gamma, 2.2×10^5 dis/min-100 cm² beta-gamma and 8.5×10^2 dis/min-100 cm² beta-gamma. No alpha contamination was detected. Gamma analysis of a chip of contaminated concrete from spot R-2 indicated the contaminant is natural uranium. Three of the spots, R-2, R-3 and R-4, exceed the allowable limits for natural uranium as found in the AEC Handbook No. N13.12 "Control of Radioactive Surface Contamination on Equipment, and Facilities to Be Released for Uncontrolled Use". These four areas indicate that the contamination is fixed and not removable. None of the four areas of contamination were found to be large areas of contamination.

The highest End Window reading of 0.09 mR/h was indicated at spot R-2. None of the other three spots indicated an elevated direct reading on the End Window Detector.

Integrated background measurements using the HPI 1010 instrument were taken at seven locations on the pad. These readings ranged from 4.9

greater than the normal background.

The concentrations of radon daughters were determined at three locations on the Building 421 pad. These were 0.00013 WL, 0.00011 WL and 0.00011 WL. According to the Surgeon General's Guidelines found in 10 CFR 20.101, concentrations of radon daughters below 0.03 WL do not require remedial action in structures other than private dwellings and schools.

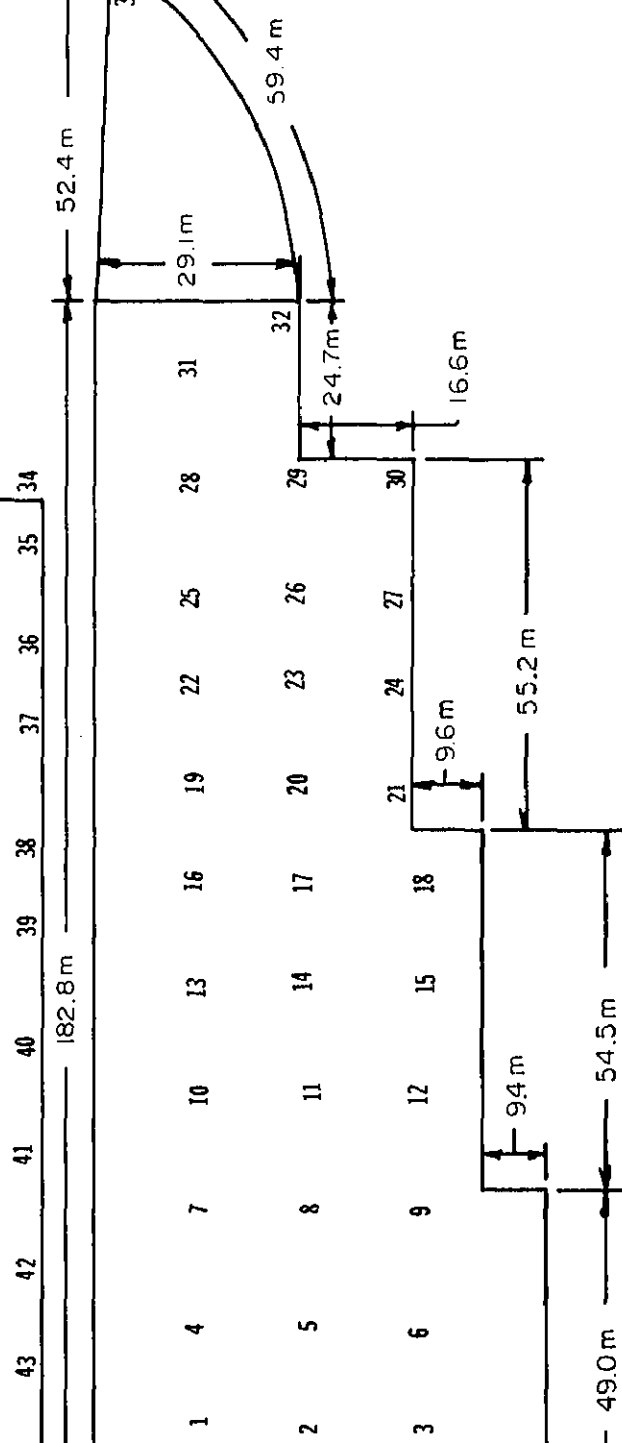
Eight soil samples were taken at locations about the site. To obtain background soil sample data, samples were collected in areas which appeared to be undisturbed since the MED/AEC era. An elevated level of uranium was detected in one background soil sample. This elevated uranium level is probably the result of the addition of phosphate fertilizer to the soil. Disregarding this sample, the background soil samples ranged from 0.9 to 4.8 pCi/g of natural uranium.

The levels of uranium found in the soil samples taken about the site ranged from 0.8 to 4.1 pCi/g. Results of the soil samples indicate elevated levels above the natural background levels in the soil.

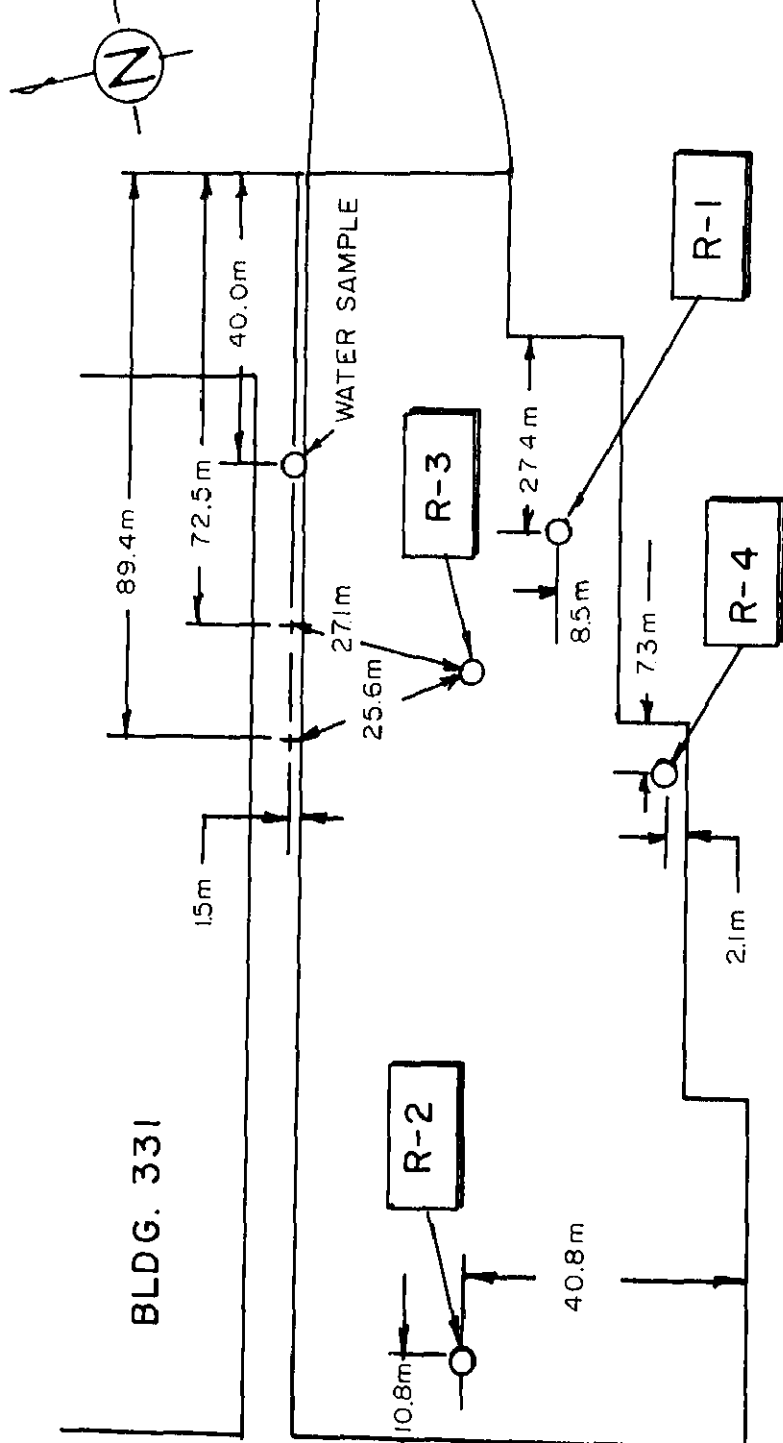
One water sample was obtained from the storm sewer line on the east side of the Building 421 pad. A gamma spectral analysis indicated no radioactivity in the sample. It is therefore felt that no contamination is present in this sewer.



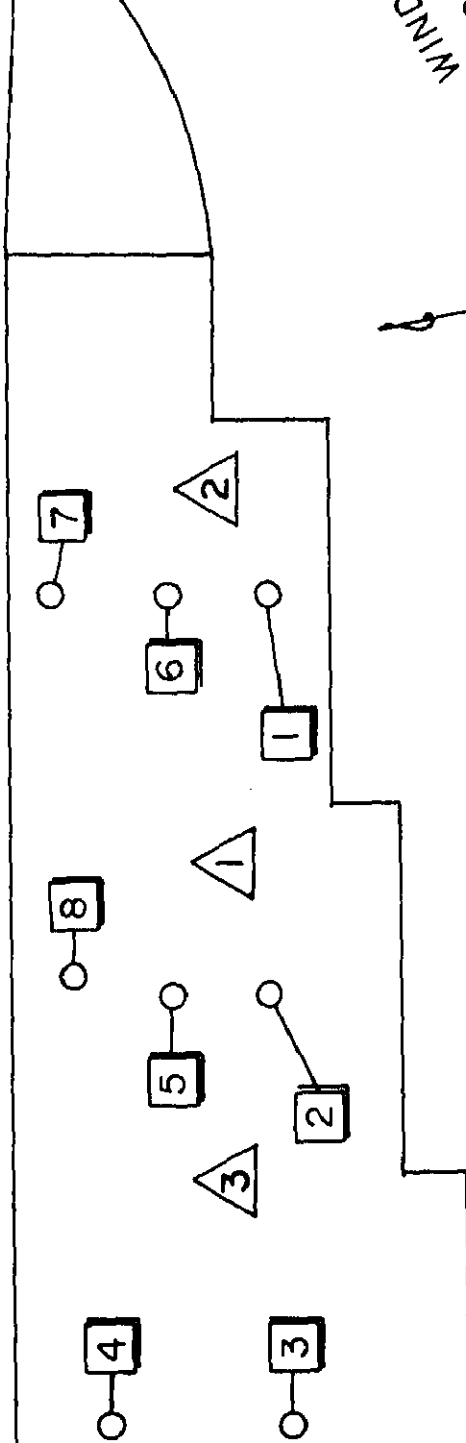
BLDG. 33 I



ANL-HP DW

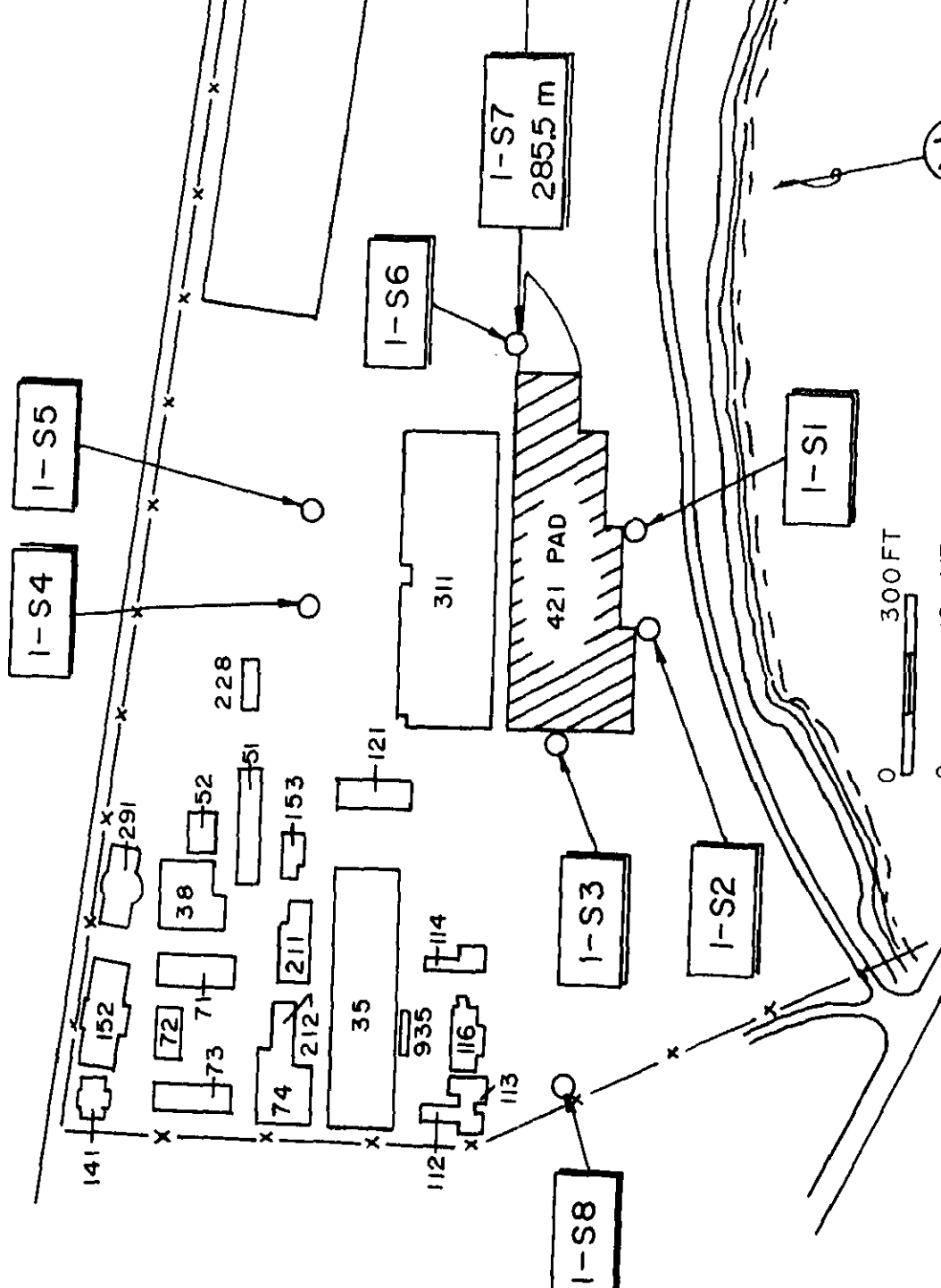


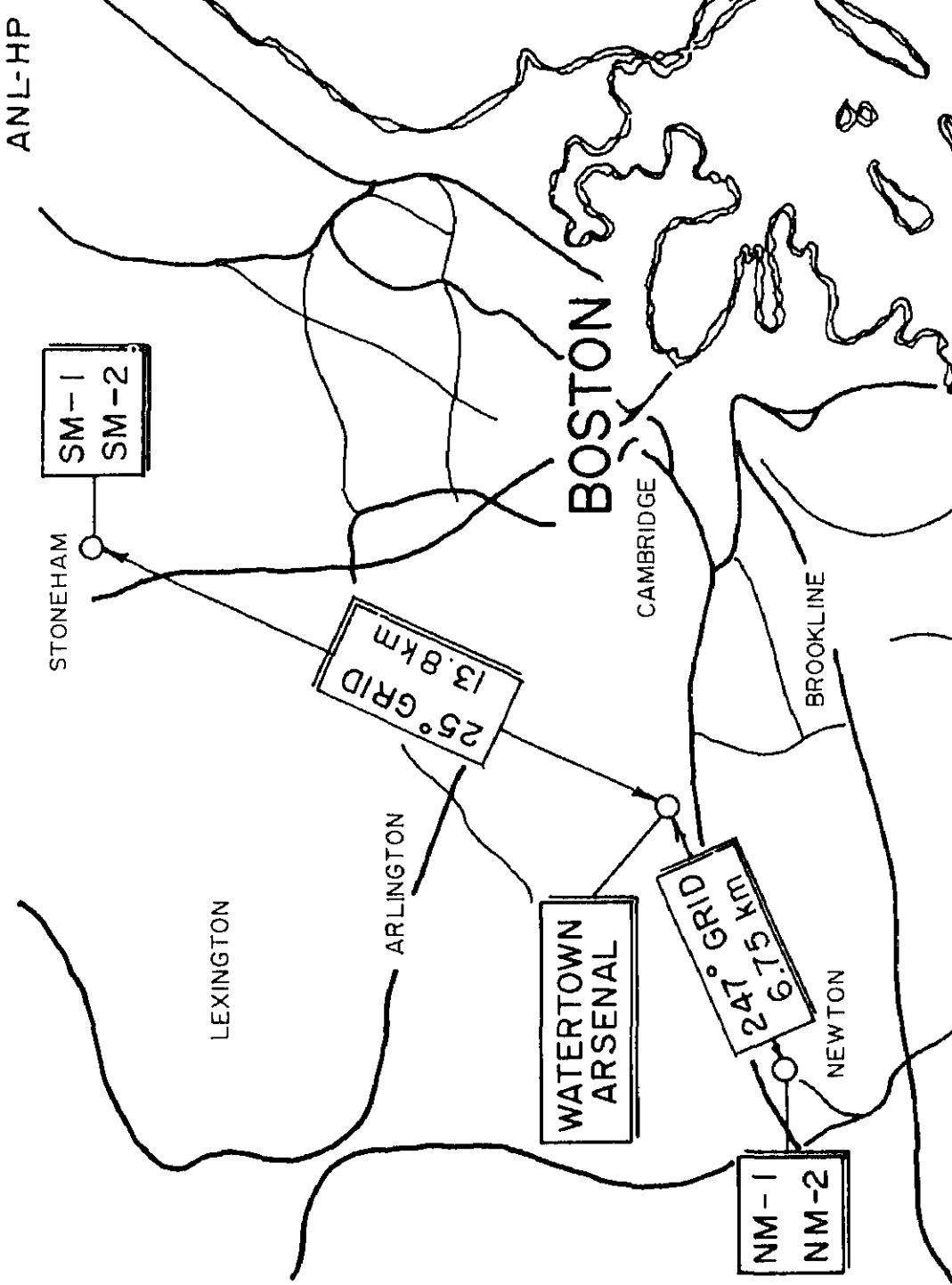
BLDG. 331



100 FT







Watertown Arsenal

Sample No. R2

Date Counted:

Time Counted:

Calibration:

Radionuclide(s):

September

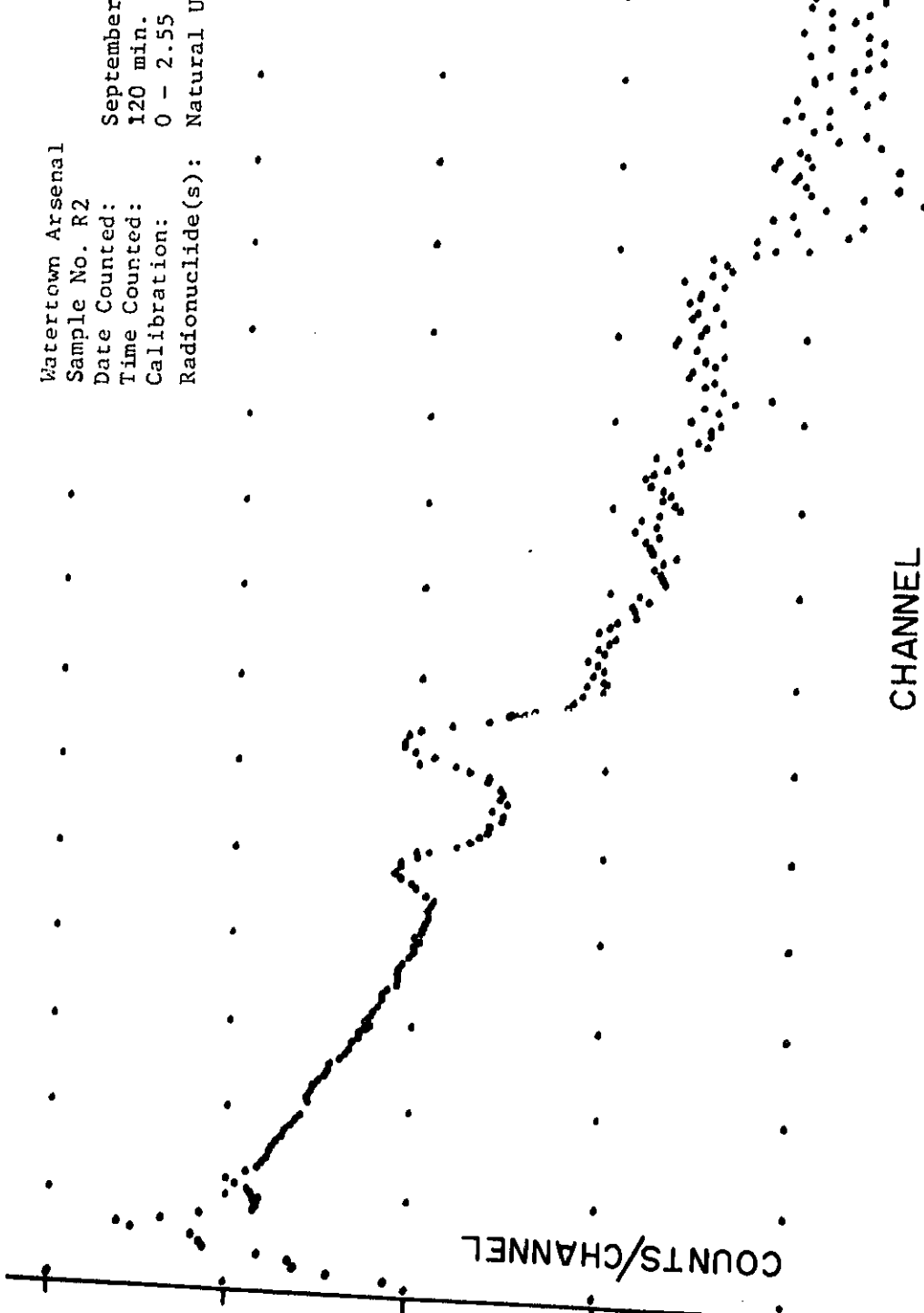
120 min.

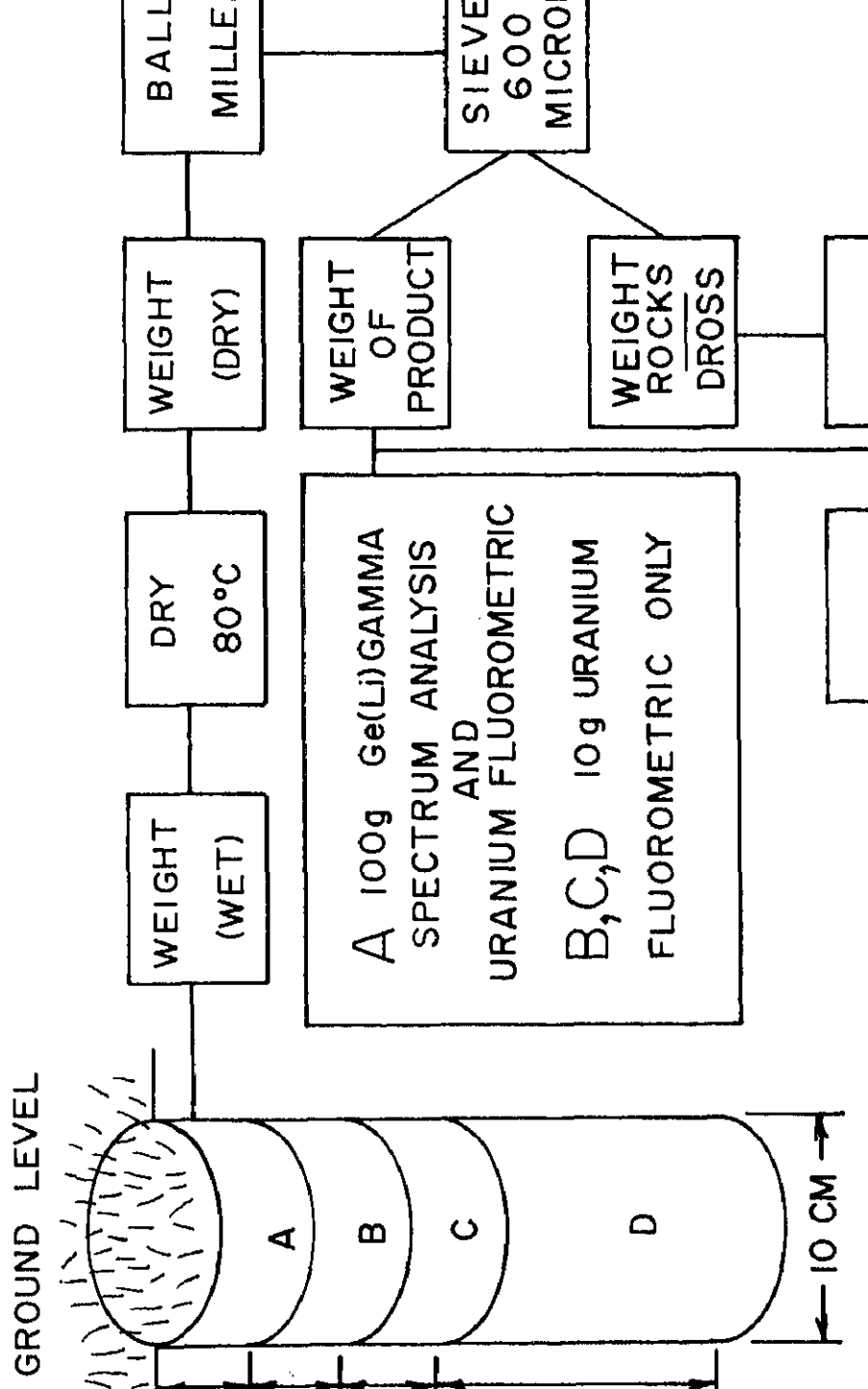
0 - 2.55

Natural U

COUNTS/CHANNEL

CHANNEL





a beta mode reading was obtained. The instrument was then switched to the alpha mode and a reading of the alpha contamination was obtained. In the alpha mode the instrument only responds to particles with high specific ionization such as alpha particles.

The beta mode readings are compensated for any alpha contribution.

- (b) BKGD (Background) The following are the instrument background readings:

	<u>Beta Mode</u>	<u>Alpha Mode</u>
Floor Monitor	1500-2000 c/min-325 cm ²	0-50 c/min-325 cm ²
PAC-4G-3	150- 200 c/min-51 cm ²	0-50 c/min-51 cm ²
PC Counter	50 c/min	1.0 c/min
10 Wire	500 c/min	10 c/min
GM End Window Detector read	0.03 to 0.05 mR/h at 1 meter above floor.	

- (c) NA (Non-applicable) No contamination was detected above background in the beta mode; therefore, no alpha mode or contact End Window Survey was necessary.

Percent of Area Accessible for Survey Floors	Air Sample (WL)	Direct Readings (a)		End Window (mR/h)		Smear Results (dis/min-100 cm ²)	Comments
		Beta	Alpha	Contact	1 Meter		
95	No walls	0.00013 0.00011 0.00009					Air Sam
							Air Sam
							Air Sam
							Spot of R-1 equ uranium
			8.4x10 ²	BKGD ^(b)	BKGD	BKGD	
			2.2x10 ⁵	BKGD	0.09	BKGD	Spot of R-2 equ uranium
			2.2x10 ⁵	BKGD	BKGD	BKGD	Spot of R-3 equ uranium
			8.5x10 ⁴	BKGD	BKGD	BKGD	Spot of R-4 equ uranium
			BKGD	NA ^(c)	NA	BKGD	Rest of

	<u>Inventory Number</u>	<u>Probe Area</u>	<u>Win Thi</u>
erline Floor Monitor -4G utilizing a PAC-4G-3	181501	325 cm ²	0.85
erline Floor Monitor -4G utilizing a PAC-4G-3	183413	325 cm ²	0.85
erline Floor Monitor -4G utilizing a PAC-4G-3	184340	325 cm ²	0.85
C-4G-3	183412	51 cm ²	0.85
C-4G-3	183414	51 cm ²	0.85
C-4G-3	183415	51 cm ²	0.85
C-4G-3	184339	51 cm ²	0.85
C-4G-3	184341	51 cm ²	0.85
erline HP-90 a-Gamma End Window	159006	-	4.1 - 4.
lear Measurement Corp. 5 2 π Internal Gas Flow nter	184065	-	0.85
onne National Laboratory Wire Flat Plate Gas portional Detector with erline Mini Scaler MS-2	184342 184343	400 cm ²	0.85
onne National Laboratory er Queen Air Sampler using 70 filter media	-	-	
th Physics Instrument el HPI-1010	184444	-	
ear Data Multichannel alyzer Model ND-100	184764	-	

INSTRUMENT BACKGROUND READINGS

Instrument	Readings*		1 meter floor
	Alpha Mode (c/min)	Beta Mode (c/min)	
1st Floor FM-4G AC-4G-3	0 - 50 0 - 50 0 - 50	1 500 - 2 000 1 500 - 2 000 1 500 - 2 000	
PAC-4G-3	0 - 50 0 - 50 0 - 50 0 - 50 0 - 50	150 - 200 150 - 200 150 - 200 150 - 200 150 - 200	
530 with Beta-Gamma window	-	-	0.0
Measurement C-5 2π Gas counter	0.3	40	
National 10-Wire Flat Gas Proportional counter with Eberline Model MS-2	10	500	
Physics Instrument PI-1010			7-12

Background readings were initially taken in the mobile laboratory and checked throughout the various areas while surveying.

<u>Number</u>	<u>Date</u>	<u>Reading*</u> <u>μR/h</u>
1	5-23-77	11.5
2	5-23-77	10.0
3	5-23-77	8.2
4	5-24-77	5.3
5	5-24-77	10.9
6	5-24-77	9.7
7	5-25-77	9.4
8	5-25-77	4.9

normal background readings range from 7 - 12 μR/h.

<u>ation</u>	<u>Date</u>	<u>dis/min-m³</u>	<u>pCi/ℓ</u>	<u>WL*</u>
	5-25-78	29	0.013	0.00013
	5-25-78	25	0.011	0.00011
	5-25-78	20	0.009	0.00009

Sample Calculation #1

$$0.013 \text{ pCi/ℓ} \cdot \frac{\text{WL}}{100 \text{ pCi/ℓ}} = 0.00013 \text{ WL}$$

Working Level (WL) is defined in 10 CFR 712 as any combination of short-lived radon daughter products in 1 liter of air that will result in the emission of 1.3×10^5 MeV of potential alpha energy. The value of the WL is derived from the alpha energy released by the total short-lived radon daughter products, RaA, RaB, and RaC' in radioactive equilibrium with 100 pCi of ^{222}Rn per liter of air.

<u>Sample No.</u>	<u>Weight (grams)</u>	<u>Weight (grams)</u>	<u>Weight (grams)</u>	<u>Weight (grams)</u>
<u>SOIL SAMPLES</u>				
1-S1-A	466.1	451.4	405.1	39
1-S1-B	998.2	970.0	921.2	43
1-S1-C	823.6	791.2	754.1	34
1-S1-D	2219.0	2047.4	1910.6	128
1-S2-A	842.5	819.2	757.4	58
1-S2-B	685.5	650.8	604.4	45
1-S2-C	993.7	937.2	861.3	74
1-S2-D	2175.6	2032.4	1586.6	431
1-S3-A	906.0	887.9	840.0	33
1-S3-B	848.6	838.1	822.8	10
1-S3-C	1045.5	1022.9	942.4	75
1-S3-D	2749.0	2676.2	1875.5	785
1-S4-A	1051.9	1040.3	536.8	499
1-S4-B	1111.0	1093.1	418.9	666
1-S4-C	2326.7	2284.3	752.3	1526
1-S4-D	5208.6	5095.2	1416.4	3557
1-S5-A	781.9	777.8	569.8	201
1-S5-B	1192.2	1175.6	451.5	720
1-S5-C	1521.8	1494.2	701.2	781
1-S5-D	3058.8	2936.9	943.3	1961
1-S6-A	637.4	626.9	395.5	220
1-S6-B	851.0	826.6	481.1	337
1-S6-C	1048.3	1007.4	537.8	463
1-S6-D	2692.6	2553.8	926.3	1619
1-S7-A	241.1	234.1	197.4	26
1-S7-B	531.8	522.6	304.8	210
1-S7-C	1653.8	1631.0	551.4	1075
1-S7-D	5387.0	5347.2	1385.4	3915
1-S8-A	352.5	289.1	262.5	23
1-S8-B	368.4	308.4	299.3	3
1-S8-C	727.1	622.0	574.7	40
1-S8-D	1529.9	1522.8	942.6	350

(cont'd)

<u>o.</u>	<u>Net Weight (grams)</u>	<u>Dry Weight (grams)</u>	<u>Sieved Weight (grams)</u>	<u>Rocks and Dross (grams)</u>
<u>BACKGROUND SOIL SAMPLES</u>				
	519.1	416.5	356.7	53.5
	668.4	588.8	334.0	247.5
	777.0	668.9	534.0	111.3
	1887.5	1622.9	1182.8	434.4
	752.5	613.9	471.7	131.7
	342.0	272.5	236.2	30.7
	983.0	814.3	699.1	108.3
	1779.4	1496.4	1271.3	211.4
	466.8	387.6	358.6	11.9
	789.9	705.4	425.4	275.0
	724.7	627.9	439.1	186.3
	1353.0	987.8	772.2	196.5
	811.1	666.6	612.2	49.5
	568.1	366.4	360.1	2.7
	487.8	331.7	324.0	3.5
	1214.4	849.6	835.0	10.6

Sample	Ge(Li) Spectra pCi/g received wt $\pm \sigma$				U
	^{137}Cs	^{232}Th Decay Chain	^{226}Ra Decay Chain	$\mu\text{g/g} \pm \sigma^{(2)}$	
A	0.09 ± 0.04	0.7 ± 0.1	0.61 ± 0.06	2.1 ± 0.4	
B				1.5 ± 0.4	
C				2.4 ± 0.9	
D				1.8 ± 0.4	
A	0.09 ± 0.04	0.6 ± 0.1	0.49 ± 0.06	1.9 ± 0.4	
B				1.5 ± 0.4	
C				1.5 ± 0.5	
D				1.3 ± 0.3	
A	0 ± 0.04	0.7 ± 0.1	0.51 ± 0.06	1.5 ± 0.4	
B				1.5 ± 0.4	
C				1.2 ± 0.3	
D				1.2 ± 0.4	
A	0.74 ± 0.04	0.8 ± 0.1	0.86 ± 0.06	2.2 ± 0.4	
B				3.1 ± 0.6	
C				3.1 ± 0.8	
D				2.1 ± 0.6	
A	2.6 ± 0.1	0.8 ± 0.1	0.86 ± 0.06	5.9 ± 0.7	
B				2.5 ± 0.3	
C				5.6 ± 0.7	
D				5.0 ± 0.9	
A	1.40 ± 0.07	0.9 ± 0.1	0.62 ± 0.06	3.1 ± 0.7	
B				2.9 ± 1.4	
C				2.5 ± 0.8	
D				3.2 ± 0.8	
A	2.7 ± 0.1	0.6 ± 0.1	0.62 ± 0.06	1.9 ± 0.7	
B				2.2 ± 0.4	
C				1.6 ± 0.4	
D				2.5 ± 0.4	

Ge (Li) Spectra pCi/g received wt $\pm \sigma^{(1)}$

^{137}Cs	^{232}Th Decay Chain	^{226}Ra Decay Chain	Uranium $\mu\text{g/g} \pm \sigma^{(2)}$	pCi/g
3.3 \pm 0.2	1.2 \pm 0.2	1.58 \pm 0.09	4.6 \pm 0.5 2.8 \pm 0.4 3.7 \pm 0.5 3.8 \pm 0.4	3.1 2.1 2.1 2.1
BACKGROUND SOIL SAMPLES				
1.25 \pm 0.06	1.2 \pm 0.1	0.96 \pm 0.06	3.4 \pm 0.5 4.3 \pm 0.6 4.1 \pm 0.6 3.1 \pm 0.4	2.1 3.1 2.1 2.1
0.91 \pm 0.05	1.5 \pm 0.1	1.00 \pm 0.07	2.5 \pm 0.5 6.8 \pm 0.6 3.5 \pm 0.5 3.2 \pm 0.4	1.1 4.1 2.1 2.1
2.2 \pm 0.1	0.9 \pm 0.1	0.68 \pm 0.06	2.5 \pm 0.4 2.2 \pm 0.4 1.3 \pm 0.4 2.4 \pm 0.4	1.1 1.1 0.1 1.1
0 \pm 0.05	0.9 \pm 0.1	0.72 \pm 0.06	4.0 \pm 0.4 17.4 \pm 0.8 5.0 \pm 0.4 3.2 \pm 0.4	2.1 12.1 3.1 2.1
0 \pm 0.4	0 \pm 0.1	0 \pm 0.06	0 \pm 0.2	

⁽¹⁾ standard deviation due to counting statistics.

⁽²⁾ results from LFE.

L conversion from Appendix 4.

Below are the conversion factors used to obtain the reaction disintegrations per minute per 100 cm² (dis/min-100 cm²).

I Conversion Factors

	<u>PAC-4G-3</u>	<u>Floor Monitor (FM-4G)</u>
To 100 cm ²	1.96	0.31
c/min to dis/min (alpha)	2	2
c/min to dis/min (beta)	2	2
c/min to dis/min (alpha from natural uranium)	3.7	3.7
c/min to dis/min (beta-gamma from natural uranium)	4.3	3.6

II Derivation of Conversion Factors

Floor Monitor (FM-4G)

Window Area: ~325 cm²

Conversion to 100 cm² = 0.31 times Floor Monitor reading.

PAC-4G-3

Window Area: ~51 cm²

Conversion to 100 cm² = 1.96 times PAC reading.

2π Internal Gas Flow Counter, PC-5

Geometry: Solid Stainless Steel Spun Top - 0.50

Geometry: Mylar Spun Top - 0.43

Mylar Spun Top Counting (double aluminized mylar window ~0.85 mg/cm²) utilizes the well of the PC-5 and is a method developed and used by the Argonne National Laboratory Health Physics Section for negating the dielectric effect in counting samples of

using a 3.2 cm x 3.2 cm x 0.3 cm natural uranium plate as a source for alpha emissions, the plate was counted in the well of a 2π ionization flow counter, PC-5, with the source leveled to an apparent 2π geometry.

The alpha c/min was found to be 27 596 c/min or $27\,596/0.50 = 5.519 \times 10^4$ dis/min.

Using the floor monitor (FM-4G) and the uranium source to convert counts per minute (c/min) to disintegrations per minute (dis/min), the count rate was found to be 15 000 c/min at contact; therefore, $5.519 \times 10^4/15\,000 = 3.67$ dis/min to c/min.

A similar reading was indicated on the PAC-4G-3, thus indicating the conversion factor for converting natural uranium c/min to dis/min from either the PAC-4G-3 or the floor monitor in the alpha mode.

The same natural uranium source, covered with 2 layers of nonconducting material, each 6.65 mg/cm² to negate the alpha emissions, was counted for composite beta and gamma emissions in the PC-5. The source was leveled to an apparent 50% geometry; however, no provision was made for backscattering.

The composite beta-gamma count was found to be 538 066 c/min or $538\,066/147.6 \times 10^6$ dis/min.

Using the FM-4G floor monitor in the beta mode and in contact with the same natural uranium source and centered on the probe, the count rate was found to be 300 000 c/min; this indicates a conversion factor of $1\,076\,000/300\,000 = 3.6$ dis/min to c/min.

Using the same covered source for the PAC-4G-3 beta mode conversion factor, the count was found to be 250 000 c/min; this indicates a conversion factor of $1\,076\,000/250\,000 = 4.3$ dis/min to c/min.

SMEAR COUNT

The conversion factors for c/min-100 cm² to dis/min-100 cm² are as follows:

1 Conversion Equation (Alpha)

$$\frac{C_n - (\text{Bkgd})}{\text{bf} \cdot \text{sa} \cdot \text{waf}} = \text{dis/min Alpha}$$

The geometry (α) of 0.43 is standard for all flat plate counting.

the self-absorption (sa) was assumed to be 1 unless otherwise determined.

If the energies of the isotopes were known, the appropriate window air factor (waf) was used; if the energies of the isotopes were unknown, the window factor for ^{239}Pu , which is 0.713, was used.

The waf for natural uranium alphas is 0.54.

II Conversion Equation (Beta)

$$\frac{\text{min} - [\text{Beta Bkgd (c/min)} + \text{Alpha c/min}]}{g \cdot bf \cdot sa \cdot waf} = \text{dis/min Beta}$$

The geometry (g) of 0.43 is standard for all flat plate counting using the sample on top.

The backscatter factor (bf) of 1.1 is used when determining beta activity in absorber media.

The self-absorption (sa) was assumed to be 1 unless otherwise determined.

If the energies of the isotopes were known, the appropriate window air factor (waf) was used; if the energies of the isotopes were unknown, the window factor for $\text{Sr-}^{90}\text{Y}$, which is 0.85 was used.

The waf for natural uranium betas is 0.85.

This appendix summarizes the air sampling calculations for radon collected using an Argonne National Laboratory designed air sampler with V-70 filter media. The attachment includes the basic assumptions and calculations used to derive the air concentrations.

Radon Concentrations Based on RaC' Results

The following postulates are assumed in deriving the Radon-222 concentrations as based on the RaC' alpha count results.

1. RaA, RaB, RaC, RaC', are in equilibrium.
2. RaA is present only in the first count and not the 100 m count.
3. One half of the radon progeny is not adhered to air bubbles, and therefore, is not collected on the filter media.
4. The geometry factor (g) is 0.43 for both the alpha and beta counts.
5. The backscatter factor (bf) of 1.0 is used for the alpha counts which is determined from RaC'.
6. The sample absorption factor (sa) for RaC' is 0.77.
7. The window air factor (waf) for RaC' is 0.8.
8. RaB and RaC being beta emitters, are not counted in the alpha mode.
9. The half-life of the radon progeny is approximately 3.8 hours based on the combined RaB and RaC half-lives.
10. No long-lived alpha emitters are present as evidenced by the alpha count.
11. For all practical purposes, RaC' decays at the rate of RaC opposite of RaB and RaC which is approximately 36 minutes.

I. Equations Used to Derive Air Concentrations

$$A_0 = \frac{A}{e^{-\lambda t}}$$

here: A_0 = Activity present at the end of the sampling
(dis/min)

A = Activity at some time interval, after end of
sampling (dis/min)

t = Time interval from end of sampling period to
counting interval

$$\lambda = \frac{0.693}{t_{1/2}}$$

$t_{1/2}$ = Half-life of isotope (min)

$$C = \frac{A_0 \lambda}{f} \cdot \frac{1}{(1 - e^{-\lambda t_s})}$$

here: C = Concentration (dis/min-m³)

A_0 = Activity on filter media at end of sampling
period (dis/min)

f = Sampling rate (m³/min = m³/h · 1h/60min)

t_s = Length of sampling time (min)

$$\lambda = \frac{0.693}{t_{1/2}}$$

$t_{1/2}$ = Half-life of isotope or controlling parent
(min)

Example Calculation - No.1 as given in Table 5.

$$A_o = \frac{15 \text{ dis/min}}{\exp \frac{-0.693 \cdot 100 \text{ min}}{36 \text{ min}}} = 102.8 \text{ dis/min}$$

$$C = \frac{102.8 \text{ dis/min} \cdot \frac{0.693}{15/60 \text{ m}^3/\text{min}}}{1 - \exp \frac{-0.693 \cdot 40 \text{ min}}{36 \text{ min}}}$$

$$= 14.7 \text{ dis/min} \cdot \text{m}^3 \cdot 2 = 29 \text{ dis/min} \cdot \text{m}^3$$

SOIL ANALYSIS PROCEDURE FOR TOTAL URANIUM
AND GAMMA-EMITTING NUCLIDES

Summary of Methods

A 60 milliliter volume of the received soil was counted in a petri dish for 10 minutes on a Ge(Li) detector over the energy range 0 - 1.5 MeV. This corresponded to between 60 to 100 g of soil, depending upon bulk soil density. All photopeaks above instrument background were converted to disintegrations per minute using a line efficiency curve based upon a National Bureau of Standards standard. The natural Thorium-232 (^{232}Th) and Radium-226 (^{226}Ra) decay chains were calculated using the 0.910 MeV Actinium-228 (^{228}Ac) and 0.420 MeV Bismuth-214 (^{214}Bi) photopeaks respectively. Cesium-137 (^{137}Cs) was used for each sample as a representative gamma emitter. Potassium-40 (^{40}K) was observed on all soil samples, as expected, but was not calculated and reported.

One gram of the soil sample was ashed and dissolved in HF-HNO_3 for total uranium analysis. A 100- μ l aliquot of the dissolved sample was mixed with 98% NaF-2% LiF and the fluorescence determined using a Jarrell-Ash fluorometer. A quenching factor was determined for each sample by using a known amount of a standard spike.

Radioactive half-lives of ^{234}U , ^{235}U and ^{238}U , as well as the abundance for each isotope were obtained as current best values from "Table of Isotopes" - 6th Edition" by C. M. Lederer, J. M. Hollander, and I. Perlman, 1967. The following values used are:

<u>Isotope</u> <u>abundance</u>	<u>Half-life (years)</u>	
^{234}U	2.47×10^5	0.005
^{235}U	7.1×10^8	0.719
^{238}U	4.51×10^9	99.276
		<u>100.001</u>

It should be noted that the abundance totals 100.0013%. Since it is not determined which isotope(s) are in error, the calculations are made assuming the 0.0013% error not accounted for.

$$\text{spA} = \lambda N = \frac{(\ln 2) N}{t_{1/2}}$$

$$\text{spA} = \text{Specific Activity}$$

$$\text{Avogadro's Number} = 6.025 \times 10^{23}$$

$$N = \text{Number of radioactive atoms per unit mass}$$

$$= \frac{\text{Avogadro's Number}}{\text{gram atomic weight}}$$

$$t_{1/2} = \text{Half-life in years (a)}$$

$$\text{spA} = \frac{0.693 \cdot 6.025 \times 10^{23}}{t_{1/2} \text{ (a)} \cdot 5.256 \times 10^5 \frac{\text{min}}{\text{a}} \cdot \text{gram atomic weight}} = \text{dis/min-gram}$$

$$\begin{aligned} \text{spA } ^{234}\text{U} &= \frac{0.693 \cdot 6.025 \times 10^{23}}{2.47 \times 10^5 \cdot 5.256 \times 10^5 \cdot 2.34 \times 10^2} \\ &= 1.374 \times 10^{10} \text{ dis/min-gram} \\ &= 1.374 \times 10^4 \text{ dis/min-}\mu\text{gram} \cdot 5.70 \times 10^{-5} \end{aligned}$$

$$= 4.76 \times 10^{-3} \text{ dis/min-}\mu\text{gram}$$

$$= 4.76 \text{ dis/min-}\mu\text{gram} \cdot 7.196 \times 10^{-3}$$

$$= 0.034 \text{ dis/min-}\mu\text{gram of natural uranium}$$

$$\text{SpA } ^{238}\text{U} = \frac{0.693 \cdot 6.025 \times 10^{23}}{4.51 \times 10^9 \cdot 5.256 \times 10^5 \cdot 2.38 \times 10^2}$$

$$= 7.4 \times 10^5 \text{ dis/min-gram}$$

$$= 0.74 \text{ dis/min-}\mu\text{gram} \cdot 9.9276 \times 10^{-1}$$

$$= 0.735 \text{ dis/min-}\mu\text{gram of natural uranium}$$

Therefore, the activity of 1 μgram of natural uranium is

$$0.783 \text{ dis/min } ^{234}\text{U} + 0.034 \text{ dis/min } ^{235}\text{U} + 0.735 \text{ dis/min}$$

$$= 1.552 \text{ dis/min-}\mu\text{gram}$$

$$= \frac{1.552 \text{ dis/min-}\mu\text{gram}}{2.22 \text{ dis/min-pCi}}$$

$$= 0.6991 \text{ pCi/}\mu\text{gram natural uranium}$$

Conversion of $\mu\text{g/g}$ to pCi/g

Example Calculation - 1-S2-A as given in Table 7.

$$[1.9 \pm 0.4] \frac{\mu\text{gram}}{\text{gram}} \cdot \frac{0.6991 \text{ pCi}}{\text{gram}} = [1.3 \pm 0.3] \frac{\text{pCi}}{\text{gram}}$$

APPENDIX 5

PERTINENT RADIOLOGICAL REGULATIONS,
STANDARDS AND GUIDELINES

Excerpts From
DRAFT AMERICAN NATIONAL STANDARD
N13.12

Control of Radioactive Surface Contamination
on Materials, Equipment, and Facilities to be
Released for Uncontrolled Use

(continued)

potentially contaminated surfaces are not accessible for measurement (e.g., inside pipes, drains, and ductwork), such property shall not be released unless it is found to be in compliance with this standard, but shall be made the subject of case-by-case evaluation.

Property shall not be released for uncontrolled use unless measurements show that total and removable contamination levels to be no greater than the values in Table 1 or Table 2. (The values in Table 2 are easier to apply when the contaminants cannot be individually identified.)

Coatings used to cover the contamination shall not be considered a solution to the contamination problem. That is, the monitoring techniques shall be used to determine, and such determination shall be made, that the total amount of contamination present on and under any coating does not exceed the values in Table 1 or Table 2 before release.

TABLE 1
SURFACE CONTAMINATION LIMITS*

Contaminants	Nuclides	Removable	Limit (Activity) (dis/min-100 cm ²)
Description	(Note 1)		Total (Fixed + Removable)
For which the occupational MPC _a is 2×10^{-13} Ci/m ³	227Ac 241, 242m, 243Am 249, 250, 251, 252Cf	20	Non- (Fixed)
For which occupational MPC _w is 2×10^{-7} Ci/m ³ or	243, 244, 245, 246, 247, 248Cm 125, 129I 237Np 231Pa 210Pb 238, 239, 240, 242, 244Pu 226, 228Ra 228, 230Th		
Nuclides not in For which the non- occupational MPC _a (Note 2) is 1×10^{-6} Ci/m ³ or less or the non-occupational (Note 4) is 1×10^{-6} Ci/m ³	254Es 256Fm 126, 131, 133I 210Po 223Ra 90Sr 232Th	200	Non-

activity in any area of 100 cm² is less than three times the limit value. For purposes of averaging with regard to isolated spots of activity, any surface or of surface shall be considered to be contaminated above the limit value applicable to 100 cm², if (1) from measurements of a representative number of sections it is determined that $1/n \sum S_i > L$, where S_i is the disintegrations per minute per cm² determined from measurement of section i; or (2) it is determined that the activity of all isolated spots or particles in any area less than 100 cm² exceeds 3L.

integrations per minute per square decimeter.

S:

Values presented here are obtained from the Code of Federal Regulations, Title 10, Part 20, April 30, 1975. The most limiting of all given values (for example, soluble versus insoluble) are to be used. In the event of the occurrence of mixtures of radionuclides, the fraction contributed by each constituent of its own limit shall be determined and the sum of the fractions shall be less than 1.

Maximum permissible concentration in air applicable to continuous exposure of members of the public as published by or derived from an authoritative source such as the National Committee on Radiation Protection and Measurements (NCRP), the International Commission on Radiological Protection (ICRP), or the Nuclear Regulatory Commission (NRC). From the Code of Federal Regulations, Title 10, Part 20, Appendix B, Table 2, Column 1.

The instrument utilized for this measurement shall be calibrated to measure at least 100 pCi of any Group-1 contaminants uniformly spread over an area of 100 cm².

Maximum permissible concentration in water applicable to members of the public.

The instrument utilized for this measurement shall be calibrated to measure at least 1 nCi of any Group-2 beta or gamma contaminants uniformly spread over an area equivalent to the sensitive area of the detector. Direct measurements for unconditional release should be performed in areas where the background is < 100 counts per minute. When the survey must be performed in areas where the background exceeding 100 counts per minute, it may be necessary to use an indirect survey method to provide the additional sensitivity required.

TABLE 2
ALTERNATE SURFACE CONTAMINATION LIMITS

(All Alpha Emitters, except U nat and Th nat, Considered as a Group)

Contamination Contingencies	Removable	Limit (Activity) (dis/min-100 cm ²) + (F Re No
contaminant cannot be identified; all alpha emitters other than U nat (Note 1) and Th nat are present; or if beta emitters comprise ²²⁷ Ac and ²²⁶ Ra	20	No
if known that all alpha emitters generated from U nat (Note 1) and Th nat; and if beta emitters are present that, while not identified, do not include ²²⁷ Ac, ¹²⁵ I, ²²⁶ Ra, and Po	200	No
if known that alpha emitters generated only from U nat (Note 1) and Th nat in equilibrium	1 000	

levels may be averaged over one square meter provided the maximum activity in any area of 100 cm² is less than three times the limit value. For purposes of averaging with regard to isolated spots of activity, any spot of surface shall be considered to be contaminated above the limit value applicable to 100 cm², if (1) from measurements of a representative number of sections it is determined that $1/\eta \sum_i S_i > L$, where S_i is the dis/min determined from measurement of section i; or (2) it is determined that the activity of all isolated spots or particles in any area less than 100 cm² exceeds 3L.

integrations per minute per square decimeter.

TES:

- (1) U nat and decay products.
- (2) The instrument utilized for this measurement shall be calibrated with a source of at least 100 pCi of any Group-1 contaminants uniformly spread over an area of 100 cm².
- (3) The instrument utilized for this measurement shall be calibrated with a source of at least 1 nCi of any Group-2 beta or gamma contaminants uniformly spread over an area equivalent to the sensitive area of the detector. Direct survey for unconditional release should be performed in areas where background is < 100 counts per minute. When the survey must be performed in an area with a background exceeding 100 counts per minute, it may be necessary to use an indirect survey method to provide the additional sensitivity required.

GUIDELINES FOR DECONTAMINATION OF FACILITIES AND EQUIPMENT
PRIOR TO RELEASE FOR UNRESTRICTED USE
OR TERMINATION OF LICENSES FOR BY-PRODUCT, SOURCE MATERIAL
OR SPECIAL NUCLEAR MATERIAL

U. S. Nuclear Regulatory Commission
Division of Fuel Cycle and
Material Safety
Washington, D.C. 20555

November 1976

the instructions in this guide in conjunction with Table I specifying radioactivity and radiation exposure rate limits which should be accomplished the decontamination and survey of surfaces or pre-equipment prior to abandonment or release for unrestricted use.

Table I do not apply to premises, equipment, or scrap containing radioactivity for which the radiological considerations pertinent to may be different. The release of such facilities or items from control will be considered on a case-by-case basis.

1. The licensee shall make a reasonable effort to eliminate residual contamination.
2. Radioactivity on equipment or surfaces shall not be covered by paint, plating, or other covering material unless contamination levels, as determined by a survey and documented, are below the limits specified in Table I prior to applying the covering. A reasonable effort must be made to minimize the contamination prior to use of any covering.
3. The radioactivity on the interior surfaces of pipes, drain lines, duct work shall be determined by making measurements at all traps, and other appropriate access points, provided that contamination at these locations is likely to be representative of contamination on the interior of the pipes, drain lines, or duct work. Surfaces of premises, equipment, or scrap which are likely to be contaminated but are of such size, construction, or location as to make the surface inaccessible for purposes of measurement shall be presumed to be contaminated in excess of the limits.
4. Upon request, the Commission may authorize a licensee to relinquish possession or control of premises, equipment, or scrap having surfaces contaminated with materials in excess of the limits specified. This may include, but would not be limited to, special circumstances such as razing of buildings, transfer of premises to another organization continuing work with radioactive materials, or conversion of facilities to a long-term storage or standby status. Such request must:

- b. Provide a detailed health and safety analysis which reflects that the residual amounts of materials on surface areas, together with other considerations such as prospective use of the premises, equipment or scrap, are unlikely to result in an unreasonable risk to the health and safety of the public.
5. Prior to release of premises for unrestricted use, the licensee shall make a comprehensive radiation survey which establishes that contamination is within the limits specified in Table 1. A copy of the survey report shall be filed with the Division of Fuel Cycle and Material Safety, USNRC, Washington, D.C. 20555, and also the Director of the Regional Office of the Office of Inspection and Enforcement, USNRC, having jurisdiction. The report should be filed at least 30 days prior to the planned date of abandonment. The survey report shall:
- a. Identify the premises.
 - b. Show that reasonable effort has been made to eliminate residual contamination.
 - c. Describe the scope of the survey and general procedures followed.
 - d. State the findings of the survey in units specified in the instruction.

review of the report, the NRC will consider visiting the facilities to be surveyed.

TABLE 1

ACCEPTABLE SURFACE CONTAMINATION LEVELS

DES ^a	AVERAGE ^b cdf	MAXIMUM ^b bdf	REMOVAL ^c
U-235, U-238, delayed decay	5 000 dis/min α -100 cm ²	15 000 dis/min α -100 cm ²	1 000 dis/min
Ac-226, Ra-226, Th-230, Th-228, U-227, I-125,	100 dis/min-100 cm ²	300 dis/min-100 cm ²	20 dis/min
Th-232, Sr-90, U-232, U-231, I-133	1 000 dis/min-100 cm ²	3 000 dis/min-100 cm ²	200 dis/min
all emitters with decay modes alpha emission (spontaneous fission) and others	5 000 dis/min $\beta\gamma$ -100 cm ²	15 000 dis/min $\beta\gamma$ -100 cm ²	1 000 dis/min

The maximum contamination level applies to an area of not more than 100 cm².

The amount of removable radioactive material per 100 cm² of surface is determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and removing the radioactive material on the wipe with an appropriate instrument of low background. When contamination on objects of less surface area is determined, the pertinent area is determined, and the entire surface should be wiped.

The average and maximum radiation levels associated with surface contamination should not exceed 0.2 mrad/h at 1 cm and 1.0 mrad/h at 5 cm. The total activity should not be more than 7 milligrams per square centimeter of total absorber.

APPENDIX 6

SURGEON GENERAL'S GUIDELINES

Part 712

Grand Junction Remedial Action Criteria

PART 712 GRAND JUNCTION
REMEDIAL ACTION CRITERIA

Purpose.

(a) The regulations in this part establish the criteria for determining the need for, priority of and selection of appropriate remedial action to limit the exposure of individuals in the area of Grand Junction, Colorado, to radiation emanating from uranium mill tailing which have been determined to be construction-related material.

(b) The regulations in this part are issued pursuant to Pub. L. 92-562 (Stat. 222) of June 16, 1972.

Scope.

The regulations in this part apply to all structures in the area of Grand Junction, Colo., under or adjacent to which uranium mill tailings have been determined to be construction-related material between January 1, 1951, and December 31, 1972, inclusive.

Definitions.

As used in this part:

(a) "Administrator" means the Administrator of Energy Research and Development or his duly authorized representative.

(b) "Area of Grand Junction, Colo.," means Mesa County, Colo.

(c) "Background" means radiation arising from cosmic rays and

(d) "ERDA" means the U.S. Energy Research and Development Administration or any duly authorized representative thereof.

(e) "Construction-related material" means any material used in the construction of a structure.

(f) "External gamma radiation level" means the average gamma radiation exposure rate for the habitable area of a structure as measured near the structure.

(g) "Indoor radon daughter concentration level" means that concentration of radon daughters determined by: (1) Averaging the results of 6 air samples taken at intervals of at least 100 hours duration, and taken at a minimum of 4 intervals throughout the year in a habitable area of a structure, or by using some other procedure approved by the Commission.

(h) "Milliroentgen" (mR) means a unit equal to one-thousandth (1/1000) of a roentgen which roentgen is defined as an exposure dose of X or gamma radiation such that the associated corpuscular emission per 0.001293 gram of air produces, in air, ions carrying one electrostatic unit of quantity of either sign.

(i) "Radiation" means the electromagnetic energy (gamma) and particulate radiation (alpha and beta) which emanate from the radioactive decay of radium and its daughter products.

(j) "Radon daughters" means the consecutive decay products of radon-222. Generally, these include Radium A (polonium-218), Radium B (lead-214), Radium C (bismuth-214), and Radium C' (polonium-214).

(k) "Remedial action" means any action taken with a reatation of reducing the radiation exposure resulting from uranigs which have been used as construction-related material in and structures in the area of Grand Junction, Colo.

(l) "Surgeon General's guidelines" means radiation guidelines relum mill tailings prepared and released by the Office of the U.S. Surgeon General, Department of Health, Education and Welfare on July 27, 1971.

(m) "Uranium mill tailings" means tailings from a uranium milling operation involved in the Federal uranium procurement program.

(n) "Working Level" (WL) means any combination of short-lived radon daughter products in 1 liter of air that will result in the ultimate emission of 1.3×10^5 MeV of potential alpha energy.

4 Interpretations.

Except as specifically authorized by the Administrator in writing, no interpretation of the meaning of the regulations in this part by an employee of ERDA other than a written interpretation by the General Counsel shall be recognized to be binding upon ERDA.

5 Communications.

Except where otherwise specified in this part, all communications concerning the regulations in this part should be addressed to the Division of Safety, Standards, and Compliance, U.S. Energy Research and Development Administration, Washington, D.C. 20545.

6 General radiation exposure level criteria for remedial action.

The basis for undertaking remedial action shall be the guidelines published by the Surgeon General of the United States. The guidelines recommend the following graded action levels for remedial actions of external gamma radiation level (EGR) and indoor radon concentration level (RDC) above background found within structures constructed on or with uranium mill tailings:

EGR	RDC	Recommendation
Greater than 0.1 mR/h	Greater than 0.05 WL	Remedial action indicated
From 0.05 to 0.1 mR/h	From 0.01 to 0.05 WL	Remedial action may be suggested
Less than 0.05 mR/h	Less than 0.01 WL	No remedial action indicated

4.7 Criteria for determination of possible need for remedial action.

Once it is determined that a possible need for remedial action exists, the owner of a structure shall be notified of that structure's eligibility for remedial action. The owner shall then engage a qualified engineering assessment to confirm the need for remedial action. The owner shall then determine the most appropriate remedial measure, if any. A determination of a possible need will be made if as a result of the presence of uranium

(a) Where ERDA approved data on indoor radon daughter concentrations are available:

(1) For dwellings and schoolrooms: An indoor radon daughter concentration level of 0.01 WL or greater above background.

(2) For other structures: An indoor radon daughter concentration level of 0.03 WL or greater above background.

(b) Where ERDA approved data on indoor radon daughter concentrations are not available:

(1) For dwellings and schoolrooms:

(i) An external gamma radiation level of 0.05 mR/h or greater above background.

(ii) An indoor radon daughter concentration level of 0.01 WL or greater above background (presumed).

(A) It may be presumed that if the external gamma radiation level is equal to or exceeds 0.02 mR/h above background, the indoor radon daughter concentration level equals or exceeds 0.01 WL above background.

(B) It should be presumed that if the external gamma radiation level is less than 0.001 mR/h above background, the indoor radon daughter concentration level is less than 0.01 WL above background, and no remedial action exists.

(C) If the external gamma radiation level is equal to or greater than 0.001 mR/h above background but is less than 0.02 mR/h above background,

(2) For other structures:

(i) An external gamma radiation level of 0.15 mR/h above background averaged on a room-by-room basis.

(ii) No presumptions shall be made on the external gamma radiation level/indoor radon daughter concentration level relationship. Decisions shall be made in individual cases based upon the results of actual measurements.

2.8 Determination of possible need for remedial action where criteria have not been met.

The possible need for remedial action may be determined where the criteria in 712.7 have not been met if various other factors are present. Such factors include, but are not necessarily limited to, size of the affected area, distribution of radiation levels in the affected area, amount of time spent in the affected area, number of individuals occupying affected area, occupancy time, and use of the affected area.

2.9 Factors to be considered in determination of order of priority for remedial action.

In determining the order or priority for execution of remedial action, due consideration shall be given, but not necessarily limited to, the following factors:

(a) Classification of structure. Dwellings and schools shall be considered first.

(b) Availability of data. Those structures for which data on

(c) Order of application. Insofar as feasible remedial action shall be given in the order in which the application is received.

(d) Magnitude of radiation level. In general, those structures with the highest radiation levels will be given primary consideration.

(e) Geographical location of structures. A group of structures in the same immediate geographical vicinity may be given priority consideration particularly where they involve similar remedial efforts.

(f) Availability of structures. An attempt will be made to schedule remedial action during those periods when remedial action can be taken with minimum interference.

(g) Climatic conditions. Climatic conditions or other seasonal considerations may affect the scheduling of certain remedial measures.

10 Selection of appropriate remedial action.

(a) Tailings will be removed from those structures where the appropriately averaged external gamma radiation level is equal to or greater than 0.05 mR/h above background in the case of dwellings and schools, and 0.1 mR/h above background in the case of other structures.

(b) Where the criterion in paragraph (a) of this section is not met, remedial action techniques, including but not limited to sealing, ventilation, and shielding may be considered in addition to that of removal. ERDA shall select the remedial action technique or combination of techniques, which it determines to be the most appropriate under the circumstances.